NON-PUBLIC?: N

ACCESSION #: 8907310022
LICENSEE EVENT REPORT (LER)

FACILITY NAME: Sequoyah Nuclear Plant, Unit 1 PAGE: 1 of 8

DOCKET NUMBER: 05000327

TITLE: Erratic feedwater controls caused a feedwater isolation on hi-hi steam generator level, which resulted in a Unit 1 reactor trip on lo-lo steam generator level.
EVENT DATE: 12/25/88 LER #: 88-047-01 REPORT DATE: 07/25/89

OPERATING MODE: 1 POWER LEVEL: 018

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR SECTION
50.73(a)(2)(iv)

LICENSEE CONTACT FOR THIS LER:
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COMPONENT FAILURE DESCRIPTION:
CAUSE: SYSTEM: COMPONENT: MANUFACTURER:
REPORTABLE NPRDS:

SUPPLEMENTAL REPORT EXPECTED: NO

ABSTRACT:

On December 26, 1988, at 0051 EST, unit 1 reactor tripped from 7-percent power on lo-lo level in S/G loop 4 following a feedwater isolation (FWI) on hi-hi level in S/G loop 2. Before this event, the main turbine was rolled at 0004 EST and reached 1700 RPM at 0037 EST. During this time, sparks were noted in the No. 10 bearing area on the turbine/generator. At 0040 EST, the shift operations supervisor elected to manually trip the turbine and then initiate a controlled reduction in power. Approximately 10 minutes later, a FWI occurred on hi-hi level in loop 2. With the reactor at approximately 22-percent power, the operator attempted to reduce reactor power to less than 5-percent to be within the capabilities of auxiliary feedwater (AFW) since main feedwater was isolated. However, as a result of a combination of the FWI, the cold AFW, and the rapid reduction in power, S/G levels dropped sharply and the unit tripped on lo-lo level in loop 4. Immediately following the trip, the balance of plant operator took manual

control of AFW to limit the reactor coolant system (RCS) cooldown in accordance with ES-0.1, "Reactor Trip Response." As average temperature (Tavg)

dropped to 540 degrees F, the lead operator started an emergency boration at 75 gpm as required. Tavg decreased to 538 degrees F before turning around and slowly increasing. The unit returned to no-load temperature and pressure and the boration was secured. The trip was a result of erratic level control on all S/G's main and bypass regulator valves for feedwater. The controllers for the bypass valves were not fine-tuned to allow for automatic operation. This forced the operator to simultaneously use manual control on the bypass and main regulator valves and consequently, resulted in the lo-lo level reactor trip. Corrective actions taken included fine-tuning the bypass valve controllers and verifying the correct operation of the main and bypass valves.

Long-term corrective actions were detailed in TVA's letter of May 5, 1989.

END OF ABSTRACT

DESCRIPTION OF EVENT

On December 25, 1988, unit 1 was taken critical in accordance with General Operating Instruction (GOI)-2 "Plant Startup From Hot Standby To Minimum Load," at 1522 EST following a forced outage on the main generator. At 1718 EST with the reactor at 1-percent power, main feedwater (MFW) pump 1A (EIIS Code SJ) was placed in service and the bypass regulator valves (EIIS Code JB) were being used to maintain steam generator (S/G) levels. Mode 1 (reactor

power greater than 5 percent) was entered at 1824 EST. Power escalation continued and the balance of plant (BOP) operator manually controlled feedwater flow on the bypass valves. At approximately 18-percent reactor power, the BOP . operator was swapping from the bypass valves to the main regulator valves (EIIS Code JB) for feedwater control. Level control was difficult and levels were varying from approximately 30 to 60-percent. During

the startup with levels being controlled using the bypass valves, Instruments Mechanics (IMs) were attempting to set the bypass controllers for automatic operation. This was not successful and was adding to the difficulty in S/G level control. At 1922 EST, a feedwater isolation (FWI) (EIIS CODE BA) start signal was generated and both motor-driven (MD) AFW pumps as well as the turbine-driven (TD) AFW pump started as a result of the FWI and subsequent MFW pump trip.

The lead operator immediately reduced reactor power to approximately 1-percent to ensure the capabilities of the AFW system were not exceeded, thereby, averting a reactor trip on lo-lo level. S/G levels dropped to 20-percent in loop 2; however, the BOP operator was able to recover S/G levels

and prevent the reactor trip. As a result of the FWI and sharp reduction in reactor power, reactor coolant system (RCS) averaged temperature (Tavg) dropped below the RCS minimum temperature for criticality (541 degrees F) to 538 degrees F and Technical Specification (TS) Limiting Condition for Operation (LCO) 3.1.1.4 action was entered at 1928 EST. The LCO was exited when Tavg recovered to greater than 541 degrees F at 1934 EST. Following the FWI and subsequent to reactor power reduction, the MFW pumps were reset and one MFW pump was returned to service. Additionally, plant management and NRC (as required by 10 CFR 50.72.b.2.ii) were promptly notified of the FWI.

Following recovery from the feedwater transient, the unit 1 assistant shift operations supervisor (ASOS) and shift operation supervisor (SOS) pulled the startup team together and discussed their actions and ways to improve S/G level control. Included in the discussion was determining the optimum use of the controllers for the bypass regulator valves. Also it was determined that the efforts by the IMs to fine-tune controllers were not beneficial and that the IMs would not be allowed to continue. Additionally, and SOS re-reviewed GOI-2, Attachment A on guidelines for MFW control during startup.

At 2040 EST, on December 25, 1988, unit 1 reentered mode 1 and increased power using the bypass valves for level control. At approximately 16-percent reactor power, the BOP operator began swapping from the bypass regulator valves to the main regulator valves.

Reactor power was increased and stabilized at 29-percent power, and at 0004 EST on December 26, 1988, the main turbine was rolled. The BOP operator was in manual control of MFW regulator valves. A vibration engineer and the turbine building (TB) ASOS were at the turbine/generator during the turbine roll for monitoring purposes.

At approximately 300 RPM in the turbine roll, a slight rubbing noise was heard at the No. 10 bearing area; however, since vibration was on the order of 3-to-4 mils, it was concluded that it was acceptable to continue the turbine roll. At approximately 1000 RPM, small sparks were seen at the No. 10 bearing oil deflector. Vibration remained in the 3-to 4-mil range. The turbine roll continued until the turbine reached the 1700 RPM point (throttle valve (TV)-governor valve (GV) transfer point). The TB ASOS was in radio contact with the main control room during this roll and he notified the SOS at 0037 EST of the sparks seen at the oil deflector. Approximately three minutes later, the SOS made the decision to manually trip the turbine based on the potential for a fire and decided to initiate a load decrease of reactor

power using the steam dump valves (EIIS Code JI). Abnormal Operating Instruction (AOI)-17 (EIIS Code JI), "Turbine and Generator Trips," was being followed by the operators during the evolution.

Approximately 10 minutes following the turbine trip, the FWI occurred as a result of hi-hi S/G level on loop 2 as the BOP operator was transferring back to the bypass regulator valves from the main regulator valves. The reactor was at approximately 22-percent power at this time. Following the FWI, rods were driven in by the lead operator in an attempt to prevent a reactor trip on lo-lo S/G level. The BOP operator attempted to control S/G levels; however, at 0051 EST, the reactor tripped on lo-lo S/G level in loop 4. Reactor power had been reduced to approximately 7-percent just before the trip.

Immediately following the trip, Emergency Procedure E-0, "Reactor Trip or SI," was followed, and after verifying no safety injection, the operators transitioned to ES-0.1, "Reactor Trip Response." The BOP operator took manual control of AFW by going to manual on the level control valves and reducing the TDAFW pump to minimum speed as directed by the ES-0.1 procedure. At five minutes into the event (0056 EST), RCS Tavg had decreased to 540 degrees F, and in accordance with ES-0.1, the lead operator started an emergency boration at a rate of 75 gpm at greater than 20,000 ppm.

The MFW pump had been reset and at approximately 0056 EST with S/G levels at greater than 18-percent in at least two S/Gs, the BOP operator shutdown the TDAFW pump.

RCS Tavg continued to decrease, and at 0101 EST, the lowest temperature of 538 degrees F was reached (based on narrow-range trend chart and postmortem printout). The boration was terminated at 0104 EST after approximately 540 gallons had been injected. RCS temperature turned around and started increasing at 0107 EST. Following the trip at 0125 EST after the unit was stabilized at no-load temperature and pressure, the RCS boron concentration was sampled and found to be 1775 ppm. The minimum boron concentration for ensuring shutdown margin at an RCS Tavg of 450 degrees F was calculated in accordance with Surveillance Instruction (SI)-38, "Shutdown Margin" following the trip and indicated that the required concentration was 1490.7 ppm. Since the actual boron concentration was 1613 ppm before the trip, shutdown margin was maintained throughout the transient.

CAUSE OF EVENT

The cause of the FWI that occurred at 1922 EST on December 25, 1989 and resulted in the start of the AFW pumps was attributed to the difficulty in controlling S/G levels. The situation was compounded by the fact that the bypass regulator valves were not "fine-tuned" to allow for automatic operation. As a result, operators were forced to use

simultaneous manual control on both the main and bypass regulator valves. Consequently, the level in S/G 2 exceeded the hi-hi level setpoint causing an FWI. The FWI signal tripped MFW pump 1A causing the AFW pumps to start.

The cause of the second FWI that resulted in an AFW pump start and ultimately in a reactor trip is also attributable to difficulty in controlling S/G levels. As was previously described, the first FWI occurred during power ascension. Conversely the second FWI occurred during power descension. Before the second isolation, Operations personnel had successfully transferred

S/G level control from the bypass regulator valves to the main regulator valves and the unit was at approximately 29-percent reactor power. While in the process of accelerating the turbine/generator up to synchronous speed, sparks were noted in the area of the number 10 bearing oil deflector. The SOS elected to manually trip the turbine for personnel safety and equipment safety reasons. Tripping the turbine in itself did not cause the FWI or the reactor trip. However, when reactor power was being decreased and Operations were attempting to transfer from the main regulator valves back to the bypass regulator valves,' the S/G level
n loop 2 exceeded the hi-hi
level setpoint causing an FWI. In contrast to the first FWI which occurred with reactor power at approximately 18-percent, the second FWI occurred with the reactor at approximately 22-percent power.

The cause of the sparks in the area of the No. 10 bearing oil deflector was determined to be that the deflector was rubbing against the turbine shaft.

Clearance measurements were taken after the turbine had coasted down and was placed on its turning gear. Those measurements supported a conclusion that rubbing would occur on the oil deflector during operation if the turbine responded in a typical manner.

A review of records made during the reassembly of the turbine/generator during

the forced outage indicated the oil deflector had been reinstalled correctly.

A contributing cause of the difficulty in controlling S/G level could be attributed to the fact that the bypass regulator valves leaked through after being closed. Acoustical testing performed after the reactor trip confirmed that the bypass regulator valves for loops 1 and 4 were both leaking through.

The leaking valves may have contributed to the erratic S/G levels.

A second contributing cause for both events is that the controllers for the bypass regulator valves are inadequately engineered in regard to human factors

considerations. When in the manual mode, the controllers are difficult to read and manipulate.

The controllers provide information on a digital display of S/G level, S/G level demand (setpoint), bypass regulator valve position, and bypass regulator

valve position demand. All parameters cannot be displayed at the same time and to check undisplayed parameters requires depressing (and maintaining depressed) a pushbutton. As a result, S/G level can be difficult to control.

ANALYSIS OF EVENT

Reactor trips, FWIs, and automatic AFW pumps starts are considered reactor protection system (RPS) and engineered safety features (ESF) actuations. Reporting of unplanned RPS and ESF actuations is required pursuant to the criteria established in 10 CFR 50.73 a.2.iv.

A review of plant conditions during this event was performed with the following conclusions:

RCS Pressure

Prior to the event, RCS pressure varied at or near 2230 psig. When the reactor trip occurred subsequent to the turbine trip, the pressurizer

pressure dropped very quickly to approximately 2145 psig. It remained at this value for less than 3 minutes. Pressure recovered to 2235 psig following the trip.

Final Safety Analysis Report (FSAR) analysis conservatively assumes that the turbine trip does not cause a reactor trip. As such, pressurizer pressure increases according to chapter 15 FSAR analysis and then falls back to lower levels following reactor trip from high pressure. The analysis further states that "reactor coolant temperatures

and pressures do not significantly increase if the steam dump system and pressurizer pressure control system are functioning properly." No pressure increase was observed, and in fact, the pressure dropped approximately 100 psi. Therefore, no challenge to the pressurizer safety

valves was posed.

The decrease in pressure can be attributed to the cooldown. Examination of the pressurizer level and pressure charts indicate both began to recover about the same time.

RCS Temperature

FSAR analysis of temperature is similar to FSAR pressure analysis. The RCS Tavg is expected to rise (as does pressure) and then drop to a post trip level following reactor trip on high RCS pressure. As before, FSAR conservatively assumed no reactor trip on turbine trip. Actual response of RCS temperature decreased to 538 degrees F as a result

of the reactor trip and post trip cooldown. As power input from the core decreased and steam dumps and the AFW system actuated, RCS Tavg dropped as expected. At the time immediately following the trip, the steam dumps were being utilized to provide a load for the reactor as the main turbine had been tripped approximately 11 minutes before the reactor trip.

The reactor trip logic for the steam dump system compares actual Tavg to a Tavg of 552 degrees F and was modified before this startup to open a designed number of dump valves upon a difference between the two values

at the time of trip. During this transient, an RCS Tave of 538 degrees F (for all 4 loops) was noted. These minimum temperatures were reached at approximately 10 minutes into the event.

Heatup/Cooldown Limits

TS limits a cooldown to a rate of 100 degrees F per hour. In this event,

RCS Tavg started at 559 degrees F at 0045 EST before the FWI and reached a minimum value of 538 degrees F at 0101 EST. Temperature returned to 547 degrees F by 0145 EST. RCS Tavg was decreasing after the FWI and had reached approximately 554 degrees F when the trip occurred. As a result the cooldown rate in this case did not exceed TS limits.

S/G Level

Several hours prior to the trip, the level in the S/Gs had experienced erratic swings as the reactor was being brought up in power and had experienced a FWI on hi-hi level in loop 2. Immediately before the trip, S/G levels dropped sharply due to the effects of the rapid power reduction and injection of cold AFW. The BOP operator took manual control to limit AFW flow and was able to control RCS cooldown. The level response was as expected based upon previous experience. No TS challenges occurred.

Pressurizer Level

Pressurizer level was increasing as a result of the power increase pretrip. Response of the pressurizer level to the transient closely paralleled that of RCS pressure.

Pressurizer level dropped rapidly as the reactor trip, AFW, and steam dumps caused the aforementioned cooldown. The pressurizer level dropped to 20-percent but did not cause the chemical and volume control system letdown flow path to isolate (setpoint is 17-percent). Subsequently, the pressurizer level returned to its program value following the trip.

Shutdown Margin

Pretrip, the reactor was operating above the insertion limits, and by definition, adequate shutdown margin was available. Following the trip, a cooldown occurred as has been previously discussed. TS require operation with a negative moderator temperature coefficient (MTC) in the cooldown transient (positive reactivity is added due to the nature of the negative MTC). RCS Tavg dropped to 538 degrees F. The required boron concentration, based upon the TS requirement of a shutdown reactivity of 1600 pcm and the most-worth rod stuck, was 1490.7 ppm at 450 degrees F.

This represents a safety margin of greater than 100 ppm (without any operator action. Therefore, TS and FSAR limits were not challenged with respect to the ability to mitigate the consequences of a concurrent steam line break at the time of minimum RCS cooldown.

Based on these analyses, the events described in this report posed no safety consequences.

CORRECTIVE ACTION

As was previously described, corrective action taken during this event by

Operations subsequent to the reactor trip was to follow Emergency Procedures E-0 and ES-0.1. To avoid an undesirable cooldown, the BOP operator took manual control of AFW as directed by ES-0.1 by transferring to manual on the level control valves and by reducing the TDAFW pump to minimum speed. Additionally, the lead operator started an emergency boration of 75 gpm when the RCS temperature decreased to 540 degrees F in accordance with ES-0.1. The BOP operator proceed to shutdown the TDAFW pump when S/G levels had recovered to at least 18-percent in two out of four loops as instructed by ES-0.1.

Corrective actions taken after the event were as follows:

1. The controllers for the bypass regulator valves were adjusted (fine-tuned) utilizing the assistance of the controller manufacturer.

2. The stroke of the main and bypass regulator valves was verified to ensure smoothness of operation, correctness of stroke, and adequate seating. Appropriate adjustments were made to the bypass regulator valves which were found to be leaking through.

3. The vendor of the turbine/generator was consulted to ascertain that the rubbing of the No. 10 bearing oil deflector on the main shaft posed no safety concerns to personnel or equipment. It was concluded that no adjustments were necessary for safe operation.

4. GOI-2 attachment A which provides guidelines for S/G level control was re-reviewed by Operations and was determined to-provide adequate

guidance to control S/G levels.

5. All work requests relating to unit 1 feedwater controls were reviewed to ensure that none could adversely affect feedwater controls for startup and power operation.

Unit 1 and unit 2 have different but comparable controllers. The operators are more familiar with the unit 2 controllers, and further the unit 2 controllers have been fine-tuned. As a result, similar problems were not anticipated during restart following the unit 2, cycle 3 refueling outage. Following the problems that did occur during restart following the Unit 2 Cycle 3 refueling outage, additional corrective actions have been identified in the areas of Operations, Maintenance, and Nuclear Engineering and will be implemented in accordance with TVA's submittal of May 5, 1989, which discussed Unit 2 reactor trips.

ADDITIONAL INFORMATION

Since initial criticality, there have been a total of 39 reactor trips (16
on unit 1 and 23 on unit 2) caused by the inability to control feedwater
flow and/or S/G level during unit startup or following system perturbations.

Since the restart of units 1 and 2 from the extended outage, this is the
first trip of either unit resulting from the inability to control S/G level.
During the extended outage before restart of the units, TVA replaced the
feedwater controllers on unit 1 and provided additional operator training.

The bypass regulator valve controllers for unit 1 were manufactured by
Turnbull.

COMMITMENTS

No additional commitments.

0498h

ATTACHMENT 1 TO 8907310022 PAGE 1 OF 1

TENNESSEE VALLEY AUTHORITY

6N 38A Lookout Place

July 25, 1989

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D.C. 20555

Gentlemen:

TENNESSEE VALLEY AUTHORITY - SEQUOYAH NUCLEAR PLANT UNIT 1 -
DOCKET NO.
50-327 - FACILITY OPERATING LICENSE DPR-77 - LICENSEE EVENT REPORT
(LER)
50-327/88047, REVISION 1

The enclosed LER is being revised to address details of long-term corrective
actions to reduce reactor trips. This event was originally reported in
accordance with 10 CFR 50.73, paragraph a.2.iv, on January 24, 1989.

Very truly yours,

TENNESSEE VALLEY AUTHORITY

J. R. Bynum, Vice President
Nuclear Power Production

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